January 26, 2007

Mr. David A. Christian
Senior Vice President and
Chief Nuclear Officer
Innsbrook Technical Center
5000 Dominion Boulevard
Glen Allen, VA 23060-6711

SUBJECT: KEWAUNEE POWER STATION, NRC EVALUATION OF CHANGES, TESTS, OR EXPERIMENTS AND PERMANENT PLANT MODIFICATIONS BASELINE INSPECTION REPORT 05000305/2006016(DRS)

Dear Mr. Christian:

On December 14, 2006, the U.S. Nuclear Regulatory Commission (NRC) completed a combined baseline inspection of the Evaluation of Changes, Tests, or Experiments and Permanent Plant Modifications at the Kewaunee Power Station. The enclosed report documents the results of the inspection which were discussed with Ms. Hartz and other members of your staff at the completion of the inspection on December 14, 2006.

The inspectors examined activities conducted under your license as they relate to safety and compliance with the Commission's Rules and Regulations, and with the conditions of your license. The inspectors reviewed selected procedures and records, observed activities, and interviewed personnel.

Based on the results of the inspection, three NRC-identified findings of very low safety significance were identified. However, because these violations were of very low safety significance, and because they were entered into your corrective action program, the NRC is treating the issues as Non-Cited Violations (NCVs) in accordance with Section VI.A.1 of the NRC's Enforcement Policy.

If you contest the subject or severity of a NCV, you should provide a response within 30 days of the date of this inspection report, with the basis for your denial, to the U. S. Nuclear Regulatory Commission, ATTN: Document Control Desk, Washington, DC 20555-0001, with a copy to the Regional Administrator, U.S. Nuclear Regulatory Commission - Region III, 2443 Warrenville Road, Suite 210, Lisle, IL 60532-4352; the Director, Office of Enforcement, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001; and the Resident Inspector Office at the Kewaunee Power Station.

In accordance with 10 CFR 2.390 of the NRC's "Rules of Practice," a copy of this letter, its enclosure, and your response (if any), will be available electronically for public inspection in the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

Sincerely,

/RA/

David E. Hills, Chief
Engineering Branch 1
Division of Reactor Safety

Docket No. 50-305
License Nos. DPR-43

cc w/encl: L. Hartz, Site Vice President
C. Funderburk, Director, Nuclear Licensing
and Operations Support
T. Breene, Manager, Nuclear Licensing
L. Cuoco, Esq., Senior Counsel
D. Zellner, Chairman, Town of Carlton
J. Kitsembel, Public Service Commission of Wisconsin
State Liaison Officer, State of Wisconsin

the NRC Public Document Room or from the Publicly Available Records (PARS) component of NRC's document system (ADAMS), accessible from the NRC Web site at http://www.nrc.gov/reading-rm/adams.html (the Public Electronic Reading Room).

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U.S. NUCLEAR REGULATORY COMMISSION

REGION III

Docket No:	50-305
License No:	DPR-43
Report No:	05000305/2006016(DRS)
Licensee:	Dominion Energy Kewaunee, Inc.
Facility:	Kewaunee Power Station
Location:	Kewaunee, WI
Dates:	October 23 through December 14, 2006
Inspectors:	R. Langstaff, Lead Inspector A. Dahbur, Reactor Inspector
Approved by:	D. Hills, Chief Engineering Branch 1 Division of Reactor Safety

SUMMARY OF FINDINGS

IR 05000305/2006016(DRS); 10/23/2006 through 12/14/2006; Kewaunee Power Station; Evaluation of Changes, Tests, or Experiments (10 CFR 50.59) and Permanent Plant Modifications.

The inspection covered a 2½-week announced baseline inspection on evaluations of changes, tests, or experiments and permanent plant modifications. The inspection was conducted by 2-region based engineering inspectors. Three Green Non-Cited Violations (NCVs) and two Unresolved Items were identified. The significance of most findings is indicated by their color (Green, White, Yellow, Red), using Inspection Manual Chapter 0609, "Significance Determination Process (SDP)." Findings for which the SDP does not apply, may be Green, or be assigned a severity level after NRC management review. The NRC's program for overseeing the safe operation of commercial nuclear power reactors is described in NUREG-1649, "Reactor Oversight Process," Revision 3; dated July 2000.

A. Inspector-Identified and Self-Revealed Findings

Cornerstone: Mitigating Systems

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action." The licensee failed to identify the impact of air intake temperature limitation on the ability of the emergency diesel generators to meet Technical Specification surveillance loading requirements at elevated temperatures. Once identified, the licensee established 75 degrees Fahrenheit as a maximum outside temperature for emergency diesel generator operability. The primary cause of this violation was related to the cross-cutting area of Problem Identification and Resolution, because the licensee failed to ensure that an issue potentially impacting nuclear safety was promptly identified, fully evaluated, and that actions were taken to address safety issues in a timely manner, commensurate with their significance.

  The issue was more than minor because the failure to identify that the emergency diesel generators would not be able to meet Technical Specification surveillance requirements at elevated temperatures could have resulted in the emergency diesel generators being considered operable when, in fact, they had less operational margin than required by Technical Specifications. The issue was of very low safety significance because both of the emergency diesel generators were determined to be capable of carrying their respective design basis accident loads below the outside temperature limitations that the licensee had in place. The issue was a NCV of 10 CFR Part 50, Appendix B, Criterion XVI, which required that conditions adverse to quality are promptly identified and corrected. (Section 1R02.1.b.2)

- Green. A finding of very low safety significance was identified by the inspectors for a violation of 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." The licensee failed to provide required fire suppression coverage in fire zone AX-32 for the safe shutdown functions of source range monitoring, isolation of a steam generator blowdown line, and pressurizer level instrumentation. Once identified, the licensee entered the issue into their corrective action program and implemented compensatory measures.

This issue was more than minor because the failure to provide suppression for redundant trains of safe shutdown equipment increased the likelihood that alternative shutdown methods would have to be used in the event of a fire. The issue was of very low safety significance because of the mitigating systems, which would have remained available in the event of a fire. The issue was a NCV of 10 CFR Part 50, Appendix R, Section III.G.3, which required fixed suppression systems for alternative shutdown areas such as fire zone AX-32. (Section 4OA5.2)

Cornerstone: Barrier Integrity

- Severity Level IV. A finding of very low safety significance was identified for the licensee's failure to adequately update the Update Safety Analysis Report (USAR) in accordance to 10 CFR 50.71, "Maintenance of Records, Making of Reports." The licensee failed to update the USAR to fully reflect changes and analyses made in response to NRC Generic Letter (GL) 96-06, "Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions." Once identified, the licensee entered this issue into their corrective action program. The primary cause of this violation was related to the cross-cutting area of Human Performance because the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee failed to provide adequate engineering procedural guidance concerning the required content of USAR updates.

  Because this issue potentially impacted the NRC's ability to perform its regulatory function, this finding was evaluated using the traditional enforcement process. The finding was determined to be more than minor because of the failure to provide complete licensing and design basis information in the USAR could result in either the licensee making an inappropriate licensing interpretation or the NRC making an inappropriate regulatory decision based on incomplete information in the USAR. The issue was of very low safety significance because no instances were identified where the failure to appropriately update the USAR impeded or influenced a regulatory decision, or resulted in an actual loss of safety function. The issue was a NCV of 10 CFR 50.71(e) which required that the USAR be updated to include the effects of all analyses of new safety issues performed by or on behalf of the licensee at Commission request. (Section 1R17.1.b.1)

B. Licensee-Identified Violations

No findings of significance were identified.

REPORT DETAILS

1. REACTOR SAFETY

Cornerstones: Initiating Events, Mitigating Systems, and Barrier Integrity

1R02 Evaluations of Changes, Tests, or Experiments (71111.02)

.1 Review of 10 CFR 50.59 Evaluations and Screenings

a. Inspection Scope

From October 23 through December 12, 2006, the inspectors reviewed one evaluation performed pursuant to 10 CFR 50.59. The inspectors confirmed that the evaluation was thorough and that prior NRC approval was obtained as appropriate. The inspectors also reviewed 16 screenings, where licensee personnel had determined that a 10 CFR 50.59 evaluation was not necessary. In regard to the changes reviewed where no 10 CFR 50.59 evaluation was performed, the inspectors verified that the changes did not meet the threshold to require a 10 CFR 50.59 evaluation. The evaluations and screenings were chosen based on risk significance, safety significance, and complexity. The list of documents reviewed by the inspectors is included as an attachment to this report.

The inspectors used, in part, Nuclear Energy Institute (NEI) 96-07, "Guidelines for 10 CFR 50.59 Implementation," Revision 1, to determine acceptability of the completed evaluations and screenings. The NEI document was endorsed by the NRC in Regulatory Guide 1.187, "Guidance for Implementation of 10 CFR 50.59, Changes, Tests, and Experiments," dated November 2000. The inspectors also consulted Part 9900 of the NRC Inspection Manual, "10 CFR Guidance for 10 CFR 50.59, Changes, Tests, and Experiments."

The above review constituted one sample for 10 CFR 50.59 evaluations and 16 samples for 10 CFR 50.59 screenings. Although the minimum sample size was not met, the inspection is considered complete because the full sample size was not available for review. Specifically, the licensee had not completed additional 10 CFR 50.59 evaluations for the period reviewed by the inspection.

b. Findings

b.1 Adequacy of 10 CFR 50.59 Screening for Procedure Change

Introduction: Based on review of 10 CFR 50.59, "Changes, Tests, and Experiments," screening 06-35-00, the inspectors identified an Unresolved Item (URI) concerning the licensee's conclusion that a 10 CFR 50.59 safety evaluation was not required. The screening was for revising procedure E-0-05 to isolate service water (SW) to the control room air conditioning (CRAC) system and other structure, system and components (SSCs).

Description: The Kewaunee Power Station Updated Safety Analysis Report (USAR) Appendix B, Section B.2.1.a indicated that Class I components and structures include those vital to safe shutdown and isolation of the reactor. USAR Table B.2.-1, “Classification of Structures, Systems And Components,” specified that the control room air conditioning and ventilation system was a Class I system. Section B.5.g.(I) indicated that Class I items were protected against damage from tornado missiles. Section B.6.1.d indicated that Class I structures were analyzed for tornado missile loads. In 2005, the licensee identified that portions of SW system lines that were located inside the Auxiliary Building fuel handling area were vulnerable to a design bases tornado missile strike. The SW lines consisted of three lines providing SW to CRAC units 1A and 1B and other SSCs. In May 2006, the licensee issued Operability Recommendation (OPR)-106, “Service Water System and Control Room Air Conditioning,” and concluded that the use of portable exhausters and opening of selected control room doors will maintain control room space temperatures below 110 degrees (°) Fahrenheit (F) following a tornado and loss of SW to the CRAC system. The licensee completed 10 CFR 50.59 screening 06-35-00 and concluded that the activity to revise procedure E-0-05 to isolate SW to the CRAC system and other SSCs following a postulated design basis tornado event did not require a 10 CFR 50.59 safety evaluation nor a license amendment.

The licensee, during the Kewaunee Power Station refueling outage in October 2006, installed a tornado missile shield to protect the subject three CRAC, SW lines per modification DCR [Design Change Request] 3628-1 and rerouted a portion of these lines per modification DCR 3628-2 to minimize the amount of piping that runs through the area of the auxiliary building that was susceptible to tornado generated missiles.

During the inspectors’ review of the licensee’s screening 06-35-00, the inspectors questioned the basis for the licensee’s conclusion that the activity to revise procedure E-0-05 did not require a 10 CFR 50.59 safety evaluation. The screening concluded that the CRAC system had no design function and therefore, had no adverse affect in the isolation of the CRAC system during a postulated design basis tornado event. The licensee’s conclusion was based on NRC Safety Evaluation Report for Kewaunee Power Station, dated May 26, 1998, which accepted the licensee’s submittal in response to Generic Letter (GL) 87-02, “Verification of Seismic Adequacy of Mechanical and Electrical Equipment In Operating Reactors (USI A-46).” The NRC indicated, in GL 87-02, that the licensee should be able to bring the plant to, and maintain, a hot shutdown condition during the first 72 hours following a safe shutdown earthquake. The licensee determined that since the NRC had accepted manual actions to open doors and set up temporary ventilation in the safety evaluation report that accepted the Kewaunee Power Station response to a seismic event, the actions would also be applicable and appropriate for tornado scenarios.

The inspectors questioned the basis for the licensee’s conclusion that the activity to revise procedure E-0-05 did not require a 10 CFR 50.59 evaluation because the licensee did not adequately evaluate the impact of the compensatory measures, which required opening the doors for the control room, on the control room post-accident recirculation system. The licensee, in 10 CFR 50.59 Screening 06-35-00, indicated that the Kewaunee Power Station licensing basis for occurrence of concurrent events was consistent with Regulatory Guide 1.117 “Tornado Design Classification,” which stated, that it was not necessary to assume a tornado event coincident with another postulated event. The licensee screening did not evaluate the affect of hazards resulted from the tornado event on the control room if the doors were to be opened as compensatory measures. Section 4.4 of NEI 96-07, which was endorsed by Regulatory Guide 1.187, indicated that if an interim compensatory action was taken to address the condition and involved a temporary procedure or facility change, 10 CFR 50.59 should be applied to the temporary change. The intent was to determine whether the temporary change/compensatory action itself (not the degraded condition) impacts other aspects of the facility or procedures described in the USAR. The Kewaunee Power Station licensing basis indicated that the purpose of the post-accident recirculation system was to isolate the control room atmosphere from hazards external to the control room, including radiation, smoke, and other airborne hazards. In addition, the inspectors determined that the manual actions described in licensee's November 10, 1995, submittal pertained to a seismic event. As such, the inspectors questioned the appropriateness of the licensee taking credit for the November 10, 1995 submittal for a tornado event. This issue is a URI pending further NRC review. (URI 05000305/2006016-01 (DRS))

b.2 <u>Failure to Identify Emergency Diesel Generator Air Intake Temperature Limitations Impact Upon Ability to Meet Technical Specification Surveillance Requirements</u>

<u>Introduction</u>: The inspectors identified an NCV of 10 CFR Part 50, Appendix B, Criterion XVI, "Corrective Action" having very low safety significance (Green) for the licensee's failure to identify a condition adverse to quality which affected the emergency diesel generators. Specifically, the licensee failed to identify the impact of air intake temperature limitations on the ability of the emergency diesel generators to meet Technical Specification surveillance loading requirements at elevated temperatures.

<u>Description</u>: At Kewanee Power Station, both emergency diesel generators were expected to exceed their continuous operating load limit of 2600 kiloWatts (kW) to meet USAR accident analysis assumptions for a loss of coolant accident with a concurrent loss off-site power. The licensee had previously determined that operation above the continuous operating limit was acceptable provided that the maximum combustion air temperatures were not exceeded and the duration was limited. On May 19, 2006, while the Kewaunee Power Station was in hot shutdown, the licensee reported, by Licensee Event Report (LER) 2006-004-00, that an incorrect interpretation of the de-rating curves for the emergency diesel generators had resulted in the potential to operate the emergency diesel generators outside the vendor recommended rating during the initial diesel loading following a design basis accident. The licensee's previous interpretation of the curves implied that it was acceptable for the emergency diesel generators to be operated in an overloaded condition under USAR accident analysis assumptions with combustion intake air temperatures as high as 115°F. Based on review by their current vendor, Engine Systems, Incorporated, the licensee determined that the previous interpretations were not correct. The previous interpretations were based on de-rating curves (supplied by a previous vendor) which were determined using a standard 90°F combustion intake air temperature. The licensee determined that the correct interpretation of the curves permitted the emergency diesel generators to be operated up to four hours per year at the loads necessary to meet USAR accident analysis assumptions, i.e., accident loads of 2945.3 kW with a maximum combustion intake air temperature of 97.5°F and 2914.2 kW with a maximum combustion intake air temperature of 101°F for the 1A and 1B emergency diesel generators, respectively. The licensee determined that, due to the design of the ventilation system for

the emergency diesel generators, combustion intake air temperatures could be significantly raised above outside air temperatures. This was due to outside air being mixed with the diesel generator room air, prior to being used for combustion air. Consequently, heat generated from the equipment in the room and the emergency diesel generator contributed towards raising combustion air intake temperatures. Measurements taken by the licensee indicated that combustion intake air temperatures could be as much as 15°F higher than outside air temperatures.

With respect to the previous incorrect interpretation, the licensee again evaluated the impact of elevated temperatures on the emergency diesel generators ability to operate under USAR accident analysis assumptions. For the 1A emergency diesel generator, the licensee removed an unnecessary load to provide greater margin. Based on their evaluation, OPR 151, "Emergency Diesel Generator 1A and 1B," the licensee determined it was appropriate to declare the emergency diesel generators inoperable when outside air temperatures of 96.8°F, for the 1A emergency diesel generator, and 99.9°F, for the 1B emergency diesel generator. The concern was that operation beyond the ratings specified on the de-rating curves would subject the emergency diesel generators to stresses beyond their designed capability. Although it wasn't expected that the additional stresses would result in an immediate catastrophic failure, exceeding the ratings would result in accelerated wear of internal engine components and could ultimately result in the inability of the emergency diesel generators to complete their mission times.

The inspectors noted that the Technical Specification surveillance requirement 4.6.a.5 specified that the emergency diesel generators be tested with a 2950 kW loading, which was higher than the loading required to meet USAR accident analysis assumptions, for two hours every operating cycle. The licensee typically performed the surveillance with a 2950 kW loading during the spring and fall seasons when outside temperatures were moderate. However, the inspectors noted that the requirement that the emergency diesel generator be able to satisfy the surveillance requirements existed anytime the emergency diesel generators were required to be operable including when elevated temperatures existed such as during the summer months. Therefore, the inspectors were concern that higher loading would restrict the operability of the emergency diesel generators to lower outside temperature than what the licensee had determined in OPR-151. Although the licensee had evaluated the impact of elevated temperatures upon diesel generator availability, i.e., the ability to operate under USAR accident analysis assumptions, the licensee had not evaluated the impact of elevated temperatures upon meeting more conservative Technical Specification surveillance requirements. During the inspection, the licensee determined that a change to their Technical Specifications would likely be required. In addition, although during the inspection the licensee was not able to determine the specific temperature for which the Technical Specification surveillance could not be performed, the licensee established 75°F as a maximum outside temperature for emergency diesel generator operability until a specific temperature could be determined or the impact of elevated temperatures upon the Technical Specification surveillance could be addressed.

<u>Analysis</u>: The inspectors determined that the failure to identify and evaluate the impact of air intake temperature limitations on the ability of the emergency diesel generators to meet technical specification surveillance loading requirements at elevated temperature was a performance deficiency warranting a significance evaluation. The inspectors concluded that the finding was greater than minor in that, if left uncorrected, the finding could have

become a more significant safety concern. Specifically, the failure to identify that the emergency diesel generators would not be able to meet Technical Specification surveillance requirements at elevated temperatures could have resulted in the emergency diesel generators being considered operable when, in fact, they could not meet the technical specification surveillance lading. Also, the failure to identify and evaluate the impact of elevated outside temperatures on the ability to meet Technical Specification surveillance requirements for operability of the emergency diesel generators affected the mitigating system cornerstone objective of ensuring the availability, reliability, and capability of systems that mitigate transients and accidents. The inspectors determined that this issue was related to the cross-cutting area of Problem Identification and Resolution because the licensee failed to ensure that an issue potentially impacting nuclear safety was promptly identified, fully evaluated, and that actions were taken to address safety issues in a timely manner, commensurate with their significance. Specifically, the licensee, in OPR-151, failed or identify the impact of air intake temperature limitations on the ability of the emergency diesel generators to meet Technical Specification surveillance loading requirements at elevated temperatures.

In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for the At-Power Situations," the inspectors performed an SDP Phase 1 screening and determined that the finding affected the Mitigation Systems Cornerstone because the finding affected the equipment performance attribute and affected the availability and reliability of both emergency diesel generators. The inspectors determined that the finding was of very low significance (Green) using IMC 0609, Appendix A, because the finding did not result in an actual loss of safety function. Both of the emergency diesel generators were determined to be capable of carrying their respective design basis accident loads below the outside temperature limitations that the licensee had in place. The licensee reviewed historic Technical Specification test data and verified that neither emergency diesel generator had been operated in the overload conditions concurrent with elevated outside air temperatures.

Enforcement: Criterion XVI of 10 CFR Part 50, Appendix B, requires, in part, that conditions adverse to quality are promptly identified and corrected. Technical Specification surveillance requirement 4.6.a.5 specified that each diesel generator shall be loaded to 2950 KW (nominal) for 2 hours every operating cycle. Contrary to the above, as of October 2006, the licensee had not identified a condition adverse to quality associated with Technical Specification surveillance requirements during elevated outside temperatures. Although the licensee had identified air intake temperature impacts on design accident loading of the emergency diesel generators, the licensee failed to recognize air intake temperature impacts on the ability to meet Technical Specification surveillance 4.6.a.5 loading requirements during elevated temperatures. Once identified, the licensee entered the finding into their corrective action program as CAP038847, "NRC Concern with EDG Surveillance Testing at Elevated Temperatures," and submitted revised LER 2006-004-01 on December 28, 2006. Because the finding was of very low safety significance and it was entered into the licensee's corrective action program (CAP038847), this violation is being treated as an NCV consistent with VI.A.1 of the NRC Enforcement Policy.
(NCV 05000305/2006016-02 (DRS))

1R17 Permanent Plant Modifications (71111.17B)

.1 Review of Permanent Plant Modifications

a. Inspection Scope

From October 23 through December 14, 2006, the inspectors reviewed eight permanent plant modifications that had been installed in the plant during the last two years. The modifications were chosen based upon risk significance, safety significance, and complexity. As per inspection procedure 71111.17B, one modification was chosen that affected the barrier integrity cornerstone. The inspectors reviewed the modifications to verify that the completed design changes were in accordance with the specified design requirements, and the licensing bases, and to confirm that the changes did not adversely affect any systems safety function. Design and post-modification testing aspects were verified to ensure the functionality of the modification, its associated system, and any support systems. The inspectors also verified that the modifications performed did not place the plant in an increased risk configuration.

The inspectors also used applicable industry standards to evaluate acceptability of the modifications. The list of modifications and other documents reviewed by the inspectors is included as an attachment to this report.

The above review constituted completion of eight samples for permanent plant modifications.

b. Findings

b.1 Failure to Fully Update Updated Safety Analysis Report

Introduction: The inspectors identified a NCV of 10 CFR 50.71, “Maintenance of records, making of reports,” having very low safety significance (Green) for the licensee’s failure to adequately update the Kewaunee Power Station USAR. Specifically, the licensee failed to update the USAR to fully reflect changes and analyses made in response to GL 96-06, “Assurance of Equipment Operability and Containment Integrity During Design-Basis Accident Conditions.”

Description: The inspectors noted that the licensee’s update of the Kewaunee Power Station Updated Safety Analysis Report for changes and analyses associated with GL 96-06 was limited. The generic letter requested that licensees evaluate potential conditions with respect to water hammer, over-pressure protection, and two-phase flow. The licensee for Kewaunee Power Station provided information to the NRC in response to GL 96-06 via several letters including letters dated January 28, 1997; November 20, 1997; and July 30, 1998.

With respect to water hammer, the licensee stated, by letter dated January 28, 1997, that they had performed a quantitative analysis of the containment fan coil unit susceptible to the most severe water hammer and a qualitative analysis of the remaining three containment fan coil units. The licensee stated in their letter, that the analyses performed determined that there was no functional concern. However, the licensee’s updating of the USAR failed to include any discussion of the containment fan coil units having been analyzed for water hammer.

With respect to over-pressure protection, the licensee stated, by letter dated November 20, 1997, that five containment penetrations had “inherent over pressure protection” in that the licensee’s evaluation determined that at least one containment isolation valve for the penetration would momentarily open and relieve pressure. However, the licensee’s updating of the USAR did not include any discussion of this evaluated “inherent” over pressure protection for these penetrations.

With respect to two-phase flow, the licensee stated, by letter dated July 30, 1998, that orifices were installed in the discharge piping of the containment fan coil units to preclude two-phase service water flow under accident conditions. As part of their response, the licensee identified the computer codes used to model flow and associated assumptions. The inspectors noted that the licensee had updated Figure 9.6-2b of the USAR, a flow diagram for the service water system, to include the orifices on the discharge lines. In addition, the licensee had updated Section 9.6.2 of the USAR to reflect the existence of the orifices with the statement that the orifices were sized to provide the required flow and tested in the safety injection line-up. However, the licensee’s updating of the USAR failed to note that in addition to ensuring adequate flow, the installed orifices were installed to ensure adequate back pressure to preclude two-phase flow. In addition, the updated USAR did not include any discussion of the computer codes and calculational assumptions used for sizing the installed orifices.

The inspectors determined that the updated information was not sufficient to permit understanding of new or modified safety analyses, design bases, and facility operation. The inspectors noted that, during the same time period as the inspection, the licensee had identified another issue with respect to the USAR not being appropriately updated. The inspectors reviewed the associated corrective action document, CAP038824, “CREZ Boundary Function Not Adequately Reflected in USAR Safety Analysis,” and determined that the corrective action had not been assigned a sufficiently high significance level to ensure that a sufficiently broad investigation would be performed to address the USAR updating process in general. Based on discussions with licensee engineering staff, the inspectors determined that although engineering procedures provided guidance with respect to when the USAR needed to be updated, the procedures did not provide sufficient guidance concerning the required content of USAR updates. In addition, based on interviews of engineering staff, the inspectors identified weaknesses in engineering staff knowledge concerning the required content for USAR updates.

<u>Analysis</u>: Because violations of 10 CFR 50.71(e) are considered to be violations that potentially impede or impact the regulatory process, they are dispositioned using the traditional enforcement process instead of the SDP. Typically, the Severity Level would be assigned after consideration of appropriate factors for the particular regulatory process violation in accordance with the NRC Enforcement Policy. However, the SDP is used, if applicable, in order to consider the associated risk significance of the finding prior to assigning a severity level. Using IMC 0612, Appendix B, “Issue Dispositioning Screening,” the inspectors determined that the finding was more than minor because of the potential to impact the regulatory process. Specifically, the failure to provide complete licensing and design basis information in the USAR could result in either the

licensee making an inappropriate licensing interpretation or the NRC making an inappropriate regulatory decision based on incomplete information in the USAR. The inspectors determined that the finding was most closely associated with the barrier integrity cornerstone of the reactor safety strategic performance area because the analyses and modifications performed in response to GL 96-06 were focused on maintaining containment integrity. The inspectors determined that this issue was related to the cross-cutting area of Human Performance because the licensee failed to ensure that personnel, equipment, procedures, and other resources were available and adequate to assure nuclear safety. Specifically, the licensee failed to provide adequate engineering procedural guidance concerning the required content of USAR updates. The inspectors determined that the finding was of very low safety significance (i.e., Green) because the inspectors did not identify any instances where the failure to appropriately update the USAR impeded or influenced a regulatory action, or resulted in an actual loss of safety function. In accordance with the Enforcement Policy, the violation was therefore classified as a Severity Level IV violation.

Enforcement: 10 CFR 50.71(e) required, in part, that licensees periodically update the Final Safety Analysis Report (FSAR) originally submitted as part of the application for the operating license to assure that the information included in the FSAR contains the latest material developed. 10 CFR 50.71(e) further required that the submittal contain all changes made in the facility and all the changes necessary to reflect information and analyses submitted to the Commission by the licensee or prepared by the licensee pursuant to Commission requirement since the submission of the original FSAR or, as appropriate, the last updated FSAR. 10 CFR 50.71(e) required, in part, that the updated FSAR be revised to include the effects of all analyses of new safety issues performed by or on behalf of the licensee at Commission request. By Generic Letter 96-06, the Commission requested analyses of a new safety issue relating to hydrodynamic effects of water hammer on containment air cooler cooling water systems, the potential for two-phase flow conditions in cooling water systems serving containment air coolers, and thermally induced overpressurization of isolated water-filled piping sections in containment. In response to Generic Letter 96-06, the licensee performed analyses relating to the hydrodynamic effects of water hammer on containment air cooler cooling water systems, the potential for two-phase flow conditions in cooling water systems serving containment air coolers, and thermally induced overpressurization of isolated water-filled piping sections in containment.

Contrary to the above, as of December 14, 2006, the licensee had not adequately updated the FSAR to reflect analyses performed relating to the effects of water hammer on containment air cooler cooling water systems, the potential for two-phase flow conditions in co cooling water systems serving containment air coolers, and thermally induced overpressurization of isolated water-filled piping sections in containment. Specifically, the licensee failed to update the FSAR to reflect that the containment fan coil units had been analyzed for water hammer, that five containment penetrations had "inherent" overpressure protection, and that orifices installed in containment fan cooler discharge piping were installed to ensure adequate back pressure to preclude two-phase flow. Once identified, the licensee entered the issue in its corrective action program as CAP039449, "USAR Not Updated to Reflect Method of Evaluation in GL 96-06, Response." Based on discussions with licensee staff, the licensee planned on revising their engineering procedures to reflect the guidance of Regulatory Guide 1.181, "Content of the Updated Final Safety Analysis Report In Accordance With 10 CFR 50.71(e)," and guidance document NEI 98-03, "Guidelines for Updating Final Safety Analysis Reports," Revision 1. (NCV 05000305/2006016-03 (DRS))

b.2 Internal Flooding Licensing Basis

Introduction: Based on review of modification DCR 3622, "RHR [residual heat removal] Pump Flooding Protection," and 10 CFR 50.59 Screening 06-23-00, the inspectors identified a URI with respect to the Kewaunee Power Station licensing basis for internal flooding.

Description: Appendix B.5, "Protection of Class I Items," of the USAR provided the criterion that "No single event will cause failure of redundant circuits or Engineered Safety Feature components in a manner such that a single failure after the event could prevent the protective functions of the associated Engineered Safety Features." In addition, USAR Appendix B.5 specified that Class I items are protected against damage from rupture of a pipe or tank resulting in serious flooding or excessive steam release to the extent that the Class I function is impaired; and earthquake, by having the ability to sustain seismic accelerations adopted for purposes of plant design without loss of function.

As part of the review of DCR 3622, "RHR [Residual Heat Removal] Pump Flooding Protection," the inspectors determined that the licensee had only assumed a single pipe failure or tank rupture for internal flooding. The inspectors noted that for seismic events, multiple failures of systems, structures, and components that had not been analyzed to withstand seismic events could, conceivably, fail and cause flooding. The inspectors believed that the assumption that only the single most limiting failure in an area would fail was a potentially non-conservative assumption.

For DCR 3622, the licensee had installed flood barriers, approximately one foot high, around the RHR pump pits covers located on the basement floor of the auxiliary building. In addition, the licensee had installed flood barriers at entrances to rooms on the floor above (i.e., the ground floor of the auxiliary building) that had floor openings above the area bounding by the flood barriers installed on the basement floor of the auxiliary building. The intent of the flood barriers installed on the ground floor level was to preclude flooding on the ground floor from affecting the RHR pumps. For example, the licensee had installed flood barriers to the RHR heat exchanger rooms because the rooms had floor openings that communicated with rooms on the floor below, i.e., the basement level, which were within the flood barriers. Any flooding into areas bounded by the flood barriers on the basement level would tend to be contained directly to the area above the RHR pumps thereby potentially affecting the RHR pumps.

The inspectors questioned whether certain potential flooding sources had been appropriately evaluated by the licensee. Specifically, a 4-inch floor drain from the "valve gallery" compartment inside the auxiliary building ground floor demineralizer room was located above the area bounded by the basement floor RHR pump pit flood barriers. Approximately six horizontal inches of the drain line for the floor drain was located above the area "protected" by the flood barriers. The licensee had evaluated the drain line as not being a significant flooding source because the drain line was normally dry.

However, the inspectors were concerned that the drain line could fail during a seismic event and cause flooding on the ground floor which would enter the area within the RHR pump pit flood barriers and adversely affect both RHR pumps. The licensee was not able to provide any documentation that indicated that the line had been evaluated to maintain its integrity during a seismic event. The inspectors noted that other potential flooding sources had been dispositioned in a similar manner. The licensee had interpreted their licensing basis as only requiring consideration of a single failure for a seismic event. As such, the licensee considered a postulated failure of the drain line to be in addition to the failure or rupture of a pipe or tank to be beyond what was required by their licensing basis. The inspectors questioned whether the licensee's interpretation was appropriate.

The inspectors identified an additional issue relating to the licensing basis for internal flooding based on review of 10 CFR 50.59 Screening 06-23-00. The subject of the screening was revision of procedure A-MDS-30 to add steps to open doors 182 and 264 and to verify doors 2, 5, 136, 263, and 268 closed for flooding event response for the safeguards alley located within the basement of the turbine and administrative building. In addition, the screening also discussed revision of procedures A-SW-02 and A-FW-05B to isolate pipe breaks in the Service Water and Auxiliary Feedwater Systems. The flooding response procedures actions were in response to a rupture from a Class I piping system such as service water. The licensee justification for these proposed changes was that the activities involved operator actions to address beyond design basis flood scenarios and conditions. The procedure changes were intended to reduce the risk associated with what the licensee considered beyond design basis events. Therefore, the licensee believed that the 10 CFR 50.59 regulations for evaluation of changes were not applicable to these procedure changes. The licensee concluded that the procedure changes for events beyond their design basis was based on their interpretation of licensing basis that Class I structures, systems, and components could not fail in a manner that would result in internal flooding. However, the inspectors noted Section B.5 of Appendix B of the Kewaunee Power Station USAR stated that Class I items were protected against damage from rupture of a pipe or tank resulting in serious flooding. The USAR did not make a distinction between flooding resulting from ruptures of Class I piping versus non-Class I piping. As such, the inspectors questioned the licensee's conclusion that protection from ruptures of Class I piping was beyond their licensing basis. In addition, the inspectors noted that the doors which the procedures directed to be closed were not qualified as flood barriers. As such, the inspectors were concerned that Class I SSCs located in the area were not protected against flood damage from a rupture of Class I piping.

The licensee, by letter dated March 17, 2006, had submitted a license amendment request to provide more specific design criteria for internal flooding evaluations. As part of the license amendment request, the licensee had requested, in part, to add the following design criteria for internal flooding evaluations:

- Only non-Class I/I* pipe or tanks are considered to fail and, of these, individual items may be determined not to fail if evaluated to withstand a Design Basis Earthquake; and

- Pipe and tank failure assume the single most limiting failure in an area, as determined by maximum flood level calculated in that area.

As such, the inspectors concluded that the appropriateness of the assumptions by the licensee will be reviewed as part of the NRC review of the license amendment request. The issue of whether the licensee's interpretation of their internal flooding licensing basis was appropriate will be considered an URI pending NRC review of the license amendment request. (URI 05000305/2006016-04(DRS))

4. OTHER ACTIVITIES (OA)

4OA2 Identification and Resolution of Problems

.1 Routine Review of Condition Reports

a. Inspection Scope

From October 23 through December 14, 2006, the inspectors reviewed five Corrective Action Process documents that identified or were related to 10 CFR 50.59 evaluations and permanent plant modifications. The inspectors reviewed these documents to evaluate the effectiveness of corrective actions related to permanent plant modifications and evaluations for changes, tests, or experiments issues. In addition, corrective action documents written on issues identified during the inspection were reviewed to verify adequate problem identification and incorporation of the problems into the corrective action system. The specific corrective action documents that were sampled and reviewed by the team are listed in the attachment to this report.

b. Findings

No findings of significance were identified.

4. **OTHER ACTIVITIES**

4OA5 Other Activities

.1 Temporary Instruction 2515/166, Pressurized Water Reactor Containment Sump Blockage (NRC Generic Letter 2004-02)

Inspection Scope:

The inspectors reviewed the licensee's activities in response to GL 2004-02, "Potential Impact of Debris Blockage on Emergency Sump Recirculation at Pressurized Water Reactors (PWRs)." The inspectors verified that changes to the facility or procedures, as described in the UFSAR, that are identified in the licensee's GL 2004-02 response were reviewed and documented in accordance with 10 CFR 50.59 and, if required, that the licensee had obtained NRC approval prior to implementing any related changes. Additionally, the inspectors reviewed elements of this modification using IP 71111.02 "Evaluation of Changes, Tests, or Experiments," and IP 71111.17, "Permanent Plant Modifications."

The inspectors verified that plant procedures had been updated to include programmatic controls that as a minimum that procedure changes made as part of the licensee's resolution of generic safety issue GSI-191 were reviewed and documented in accordance with 10 CFR 50.59.

Findings and Observations:

At the time of the inspection, the inspectors found that the licensee had not yet updated its licensing bases to reflect the corrective actions taken in response to GL 2004-02. The inspectors reviewed the licensee's draft USAR Change Request R20-069 for design change request DCR 3605. The inspectors noted that although the draft change request described some of the physical changes associated with the sump modification, the draft change request did not discuss the modified sump was designed to be fully submerged to preclude air entrainment as discussed in licensing correspondence for the sump. The inspectors considered the full submergence design to be a significant design feature. In addition, the draft change request did not discuss the methodology used for analysis of the modified sump even though the analysis methods were described in licensing correspondence. The inspectors noted that the draft change request would not have satisfied the requirements of 10 CFR 50.71(e). However, as the change request was still draft and had not yet been reviewed and approved by licensee management, no violation of NRC requirements had yet occurred. The licensee initiated CAP038857, "USAR Revision for DCR 3605," to address the concerns raised by the inspectors. Section 1R17.1.b.1 of this report discusses another NRC identified issue with respect to updating the USAR in accordance with 10 CFR 50.71(e).

.2 (Closed) Unresolved Item 05000305/2004005-09: Acceptability of cable spreading area suppression system.

Introduction: The inspectors identified a NCV of 10 CFR Part 50, Appendix R, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979," having very low safety significance (Green) for the licensee's failure to provide adequate fire suppression coverage for fire zone AX-32. Specifically, the licensee failed to provide required fire suppression coverage for safe shutdown functions of source range monitoring, isolation of a steam generator blowdown line, and pressurizer level instrumentation.

Description: The cable spreading area in fire zone AX-32 contained safe shutdown cables from both trains. The requirements of 10 CFR Part 50, Appendix R, III.G.3 were applicable to the area. The cable spreading area was open to the materials storage area and main feedwater and main steam isolation valves on one end. The walls and the ceiling of the cable spreading area consisted of concrete. However, the flooring of the cable spreading area consisted of 20-gauge metal decking. The radiation protection office and associated facilities were located directly below the cable spreading area and presented a fire hazard for the cable spreading area. (See Section 1R05.10.6 of Inspection Report 05000305/2004005(DRS) for further discussion of the fire hazards present and potential impact upon the cable spreading area.)

The fixed suppression system installed by the licensee consisted of a wet-pipe water spray system designed to protect a number of cable trays in the cable spreading area. The original hydraulic calculation for the cable spreading area suppression system, Calculation C10059, "Aux Bldg. Cable Trays No.'s 1AT16S5; 1AT17S6; 1AT14S6" identified that the suppression system was only designed to provide suppression for cable trays 1AT16S5, 1AT17S6, and 1AT14S6. Field walkdowns conducted by the inspectors confirmed that the suppression system was installed for these cable trays and that suppression was generally not provided for other cable trays in the cable spreading area. None of the cable trays in the cable spreading area had credited fire barriers (such as fire wrap). As part of the original Appendix R reviews, the NRC had required that a suppression system be provided for the area. However, the NRC had not reviewed suppression provided for specific cable trays.

The inspectors identified three safe shutdown functions with redundant trains of equipment located in fire zone AX-32 which lacked fire suppression. Specifically, the licensee had not adequately addressed the lack of suppression for source range flux monitoring, isolation of the "B" steam generator blowdown line, and pressurizer level indication. In discussions with licensee engineering staff, the licensee believed that the alternative shutdown capability, which was independent of fire zone AX-32, and which was used to satisfy the requirements of 10 CFR Part 50, Appendix R, Section III.L, could be considered the redundant train. However, the alternative shutdown capability specified required actions to be performed by operators either locally or at the remote shutdown panel. As such, although the shutdown methods satisfied 10 CFR Part 50, Appendix R, Section III.L, the shutdown methods did not qualify as a redundant train free of fire damage. Consequently, the requirements of 10 CFR Part 50, Appendix R, Section III.G were applicable, including the requirement to provided a fixed suppression system.

Cable routing information provided by the licensee indicated the cables for the “A” train source range monitor, SRM-28044, were routed in cable tray RED-8; and the cables for the “B” train source range monitor, SRM-28038, were routed in cable tray WHT-9. Suppression was not provided for cable trays RED-8 and WHT-9 located within the cable spreading area. Source range neutron monitoring instrumentation was required for hot shutdown to verify that the reactor was shutdown. The inspectors noted that the supplemental Safety Evaluation Report for fire protection, dated December 22, 1981, specifically required that a source range flux monitor be added to the remote shutdown panel to provide an immediate indication of a potential positive excursion from low reactivity within the reactor core. The licensee did provide source range monitoring at the remote shutdown panel in order to satisfy 10 CFR Part 50, Section III.L. However, such monitoring capability required the use of an operator outside of the control room and communications between the control room and the remote shutdown panel. As such, source range monitoring capability at the remote shutdown panel did not qualify as a redundant train free of fire damage.

The blowdown isolation valve for the “B” steam generator located outside containment, valve BT-3B, was located within fire area AX-32. Cable routing information provided by the licensee indicated that the cables for the redundant valve, valve BT-2B located within containment, was routed through cable trays 1AL6S6, 1AT16S6, 1AT18S6, 1AT19S6, 1AT20S6, and 1AT22S6 which did not have suppression. Isolation of the “B” steam generator was required for hot shutdown to ensure that the steam generator did not reach a dry out condition. Procedure E-FP-08, Appendix C, specified manual actions to isolate the “B” steam generator blowdown line by closing manual valves locally which was acceptable for meeting 10 CFR Part 50, Appendix R, Section III.L, but did not qualify as a redundant train free of fire damage.

Cable routing information provided by the licensee indicated that the cables for both channels which provided pressurizer level indication to the control room (i.e., cables for level transmitters LT-24029 and LT-24031) were routed through fire zone AX-32. A cable for level transmitter LT-24029 was routed through cable tray 1AX13N and a cable for level transmitter LT-24031 was routed through cable tray WHT-9. Suppression was not provided for either cable tray. Procedure E-FP-08 identified that redundant pressurizer level instruments were routed through fire zone AX-32 and that alternate pressurizer level indication instrumentation was provided at the remote shutdown panel. However, such monitoring pressurizer level indication at the remote shutdown panel required the use of an operator outside of the control room and communications between the control room and the remote shutdown panel. As such, monitoring at the remote shutdown panel did not qualify as a redundant train free of fire damage.

The inspectors noted that GL 86-10, “Implementation of Fire Protection Requirements,” dated April 24, 1986, clarified that suppression systems needed to be sufficient to protect against the hazards of the area. The licensee had performed an evaluation, documented in evaluation FPEE-049, “Evaluation of Partial Area Suppression/Detection,” to justify the existing partial suppression system as being adequate. However, the licensee’s evaluation failed to address the fire hazard presented by the radiation protection office below. The inspectors noted that, due to the lack of a fire barrier between the radiation protection office and the cable spreading area, a significant fire in the radiation protection office result in high temperatures in the cable spreading area. In addition, evaluation

FPEE-049 documented calculations intended to show that a localized fire in the cable spreading area could not adversely affect other trains of equipment in the area. However, while the calculations considered a transient combustible as a fire source, the calculations did not account for fire propagation to other combustibles. For example, a transient combustible fire could result in ignition of a cable tray close to the floor which could then result in fire progression to cable trays above. In addition, the calculations did not account for radiant heat flux upon nearby cable trays. As such, evaluation FPEE-049 failed to adequately address the hazards present from either a localized fire or a significant from fire originated in the radiation protection office area.

Analysis: In accordance with IMC 0612, "Power Reactor Inspection Reports," dated November 2, 2006, the inspectors determined that the issue of failing to provide adequate suppression was a performance deficiency. This performance deficiency was determined to be greater than minor because it affected the mitigating systems cornerstone attribute of protection against external factors (fire). Specifically, the failure to provide suppression for redundant trains of safe shutdown equipment increased the likelihood that alternative shutdown methods would have to be used in the event of a fire. In accordance with IMC 0609, Appendix A, "Significance Determination of Reactor Inspection Findings for At-Power Situations," dated, November 22, 2006, the inspectors determined that a significance determination using IMC 0609, Appendix F, "Fire Protection Significance Determination Process," dated February 28, 2005, was required. The inspectors performed an SDP Phase 1 screening and determined that the finding affected the Fire Prevention and Fixed Fire Protection Systems Category with a high degradation rating in accordance with IMC 609, Appendix F, Attachment 2, "Degradation Rating Guidance Specific to Various Fire Protection Program Elements," dated February 28, 2005. The inspectors performed a Phase 2 analysis in accordance with IMC 0609, Appendix F.

Step 2.9 of IMC 0609, Appendix F, outlined the formula for determining safety significance as follows:

$$\Delta CDF \approx DF \times F \times SF \times AF \times PNS \times CCDP$$

where

DF is the duration factor. For this case, the inspectors determined that DF = 1.0 based on review of task 1.4.1 of IMC 609, Appendix F, because the condition had existed for greater than 30 days.

F is the fire frequency. For this case, the inspectors determined that $F = 1.7 \times 10^{-3}$ based on review of IMC 609, Appendix F, Attachment 4, "Fire Ignition Source Mapping Information: Fire Frequency, Counting Instructions, Applicable Fire Severity Characteristics, and Applicable Manual Fire Suppression Curves," dated February 28, 2005, based on the radiation protection office having a high amount of transient combustibles.

SF is the severity factor. For this case, the inspectors conservatively assumed that SF = 1.0 because the team did not develop a fire scenario for the issue.

AF is the ignition source specific frequency adjustment factor. For this case, the inspectors determined that AF = 1.0 based on review of task 2.4.2 of IMC 609, Appendix F, and that the finding was not related to fire protection program administrative controls.

PNS is the probability of non-suppression. For this case, the inspectors conservatively assumed that PNS = 1.0 because the team did not develop a fire scenario and did not credit fire suppression activities. It was the inspectors judgment that PNS would have been much less than 1.0 had a fire scenario been developed.

CCDP is the conditional core damage probability. For this case, the inspectors reviewed the risk-informed notebook for the Kewaunee Power Station and determined that the transients without power conversion system and stuck-open power operated relief valve (PORV) significance determination process worksheets were applicable. Based on review of cable location information provided by the licensee for fire area AX-32, the inspectors determined that auxiliary feedwater would not be affected, one train of safety injection could be affected, residual heat removal could be affected (however, one train could be recovered), and one of the pressurizer PORV valves and the opposite train pressurizer PORV block valve could be affected. In addition, based on review of Procedure E-FP-08 and interviews of licensed operators during previous inspections, the inspectors concluded that it would be improbable that operators would either perform alternative shutdown from outside the control room or de-energize one of the safety-related buses due to a fire in fire area AX-32. The inspectors evaluated the amount of mitigation credit using the transients without power conversion system and stuck-open PORV significance process worksheets. For the transients without power conversion system worksheet, the inspectors concluded that a minimum of four points of recovery credit would be available due to auxiliary feedwater. In addition, the inspectors noted that high pressure recirculation could be recovered and that one train of high pressure injection (safety injection) would be available. For the stuck-open PORV worksheet, the inspectors concluded that a minimum of four points of recovery credit would be available due to the block valve for the affected PORV remaining available, recovery of low pressure recirculation, recovery of high pressure recirculation, one train of high pressure injection would remain available, and auxiliary feedwater would be unaffected. As such, the inspectors determined that CCDP = 1×10^{-4} due to the four points of mitigation credit.

ΔCDF is the difference in the core damage frequency due to the performance deficiency. For this case, the inspectors determined that the ΔCDF $\approx 1.7 \times 10^{-7}$ based on the formula and values described above.

Based on review of Table 2.9.1 of IMC 609, Appendix F, the inspectors determined that the issue of not providing adequate fire suppression was of very low safety significance (Green) because the ΔCDF of approximately 1.7×10^{-7} was less than 1×10^{-6}.

Enforcement: 10 CFR 50.48(a)(1) required each operating nuclear power plant to have a fire protection plan that satisfied 10 CFR Part 50, Appendix A, Criterion 3. 10 CFR 50.48(b) specified that 10 CFR Part 50, Appendix R established the requirements necessary to meet 10 CFR Part 50, Appendix A, Criteria 3 for nuclear power plants licensed to operate before January 1, 1979. Kewaunee Power Station, an operating nuclear power plant, was licensed to operate before January 1, 1979. 10 CFR Part 50, Appendix R, Section III.G.3 specified, in part, that alternative of dedicated shutdown capability and its associated circuits, independent of cables, systems or components in the area, room, zone under consideration should be provided: where the protection of systems whose function is required for hot shutdown does not satisfy the requirement of 10 CFR Part 50, Appendix R, Section III.G.2; or where redundant trains of systems required for hot shutdown located in the same fire area may be subject to damage from fire suppression activities or from the rupture or inadvertent operation of fire suppression systems. In addition, 10 CFR Part 50, Appendix R, Section III.G.3 required, in part, that fire detection and a fixed fire suppression system shall be installed in the area, room, or zone under consideration. Fire zone AX-32 of Kewaunee Power Station, a fire zone located outside of primary containment, had cables or equipment, including associated non-safety circuits that could prevent operation or cause maloperation due to hot shorts, open circuits, or shorts to ground, of redundant trains of systems necessary to achieve and maintain hot shutdown conditions are located within the same fire area. Specifically, fire zone AX-32 contained redundant cables or equipment for source range neutron flux monitoring, isolation of the “B” steam generator blowdown line, and pressurizer level indication. The fire protection features of fire zone AX-32 did not meet the requirements of 10 CFR Part 50, Appendix R, Section III.G.2 in that redundant trains were not separated by a distance of more than 20 feet with no intervening combustibles or fire hazards and lacked fire barriers. As such, the requirements of 10 CFR Part 50, Appendix R, Section III.G.3 were applicable to Fire Zone AX-32.

Contrary to the above, as of December 14, 2006, Fire Zone AX-32 did not have a fixed fire suppression system installed to provide fire suppression for redundant cables or equipment located within Fire Zone AX-32 for source range neutron flux monitoring, isolation of the “B” steam generator blowdown line, and pressurizer level indication. Once identified, the licensee entered the finding into their corrective action program as CAP040096, “CSR Fire Suppression System Coverage - NRC Potential NCV of Appendix R, III.G.3.” dated December 15, 2006, and implemented compensatory measures. Because the finding was of very low safety significance and it was entered into the licensee's corrective action program (CAP040096), this violation is being treated as an NCV consistent with VI.A.1 of the NRC Enforcement Policy. (NCV 05000305/2006016-05 (DRS)) Unresolved Item 05000305/2004005-09(DRS) is considered closed.

4OA6 Meetings

.1 Exit Meeting

The inspectors presented the inspection results to Ms. Hartz and others of the licensee’s staff, on December 14, 2006. Licensee personnel acknowledged the inspection results presented. Proprietary information was reviewed during the inspection and was handled in accordance with NRC policy.

ATTACHMENT: SUPPLEMENTAL INFORMATION

SUPPLEMENTAL INFORMATION

KEY POINTS OF CONTACT

Licensee
L. Hartz, Site Vice-President
L. Armstrong, Director, Engineering
T. Breene, Manager, Nuclear Licensing
K. Davison, Director, Operations and Maintenance
D. Lohman, Manager, Design Engineering
M. Sortwell, Lead, Independent Review Group
T. Webb, Director, Safety and Licensing

Nuclear Regulatory Commission
S. Burton, Senior Resident Inspector

ITEMS OPENED, CLOSED, AND DISCUSSED

Opened

05000305/2006016-01	URI	Adequacy of 10 CFR 50.59 Screening for Procedure Change
05000305/2006016-04	URI	Internal Flooding Licensing Basis

Opened and Closed

05000305/2006016-02	NCV	Failure to Identify Emergency Diesel Generator Air Intake Temperature Limitations Impact Upon Ability to Meet Technical Specification Surveillance Requirements
05000305/2006016-03	NCV	Failure to Fully Update Updated Safety Analysis Report
05000305/2006016-05	NCV	Failure to Provide Suppression for Safe Shutdown Equipment in Appendix R, III.G.3 Area

Discussed

None.

LIST OF DOCUMENTS REVIEWED

The following is a list of licensee documents reviewed during the inspection, including documents prepared by others for the licensee. Inclusion on this list does not imply that NRC inspectors reviewed the documents in their entirety, but rather, that selected sections or portions of the documents were evaluated as part of the overall inspection effort. Inclusion of a document in this list does not imply NRC acceptance of the document, unless specifically stated in the inspection report.

IR02 Evaluation of Changes, Tests, or Experiments (71111.02)

10 CFR 50.59 Screenings

05-041; DCR 3566; Revision 0

06-002; DCR 3556; Revision 0

06-007; DCR 3605; Revision 2

06-018; DCR 3622; Revision 1

06-023; Revise Procedure A-MDS-30, A-SW-02 and A-FW-05B; Revision 0

06-034; Revise Procedure N-BT-07A, N-SFP-21, N-LWP-32A, N-CVC-35M; Revision 0

06-035; Procedure E-0-05 Revision T; Revision 0

06-044; OPR 151 Compensatory Measures for Removing Load on Diesel Generator “A”; Revision 0

06-052; DCR 3597; Revision 0

06-053; Temporary Modification 2006-04; Revision 0

06-061; DCR 3628-1 Install Tornado Missile Shield to Protect Control Room Air Conditioning (CRAC) Units Service Water Lines; Revision 0

06-068; Sequence Events Recorder (SER) Points 304, 360, 361 and 476 Disabled; Revision 0

PTE 96-0001; Procurement Technical Evaluation - Fan Motor Mounting Bolt Replacement; Revision 7

PTE 01-0041; Consolidated Relief Valves; Revision 5

PTE 06-0010; Procurement Technical Evaluation - Flange Upstream of Valve SW-30B1 Replacement; Revision 0

PTE 06-0031; EDG fuel Oil System; Revision 2

10 CFR 50.59 Evaluations

06-02-001; TRM 3.11.1 Core Surveillance Instrumentation(Incore Thimbles); dated August 18, 2006

IR17 Permanent Plant Modifications (71111.17B)

Modifications

DCR 3556; Safety Injection Accumulator Level Transmitter Replacement; Revision 0

DCR 3605; Replacement of the ECCS Sump B Strainer; Revision 3

DCR 3622; RHR Pump Flooding Protection; Revision 1

DCR 3628-1; Install Tornado Missile Shield to Protect Control Room Air Conditioning (CRAC) Units Service Water Lines; Revision 0

PTE 96-0001; Procurement Technical Evaluation - Fan Motor Mounting Bolt Replacement; Revision 7

PTE 01-0041; Consolidated Relief Valves; Revision 5

PTE 06-0010; Procurement Technical Evaluation - Flange Upstream of Valve SW-30B1 Replacement; Revision 0

PTE 06-0031; EDG Fuel Oil System; Revision 2

Other Documents Reviewed During Inspection

Corrective Action Program Documents Generated As a Result of Inspection

CAP038847; NRC Concern with EDG Surveillance Testing at Elevated Temperature; dated October 26, 2006

CAP038849; Portable Crane for Use with RHR Pump Pit Blocks; dated October 26, 2006

CAP038857; USAR Revision for DCR 3605; dated October 27, 2006

CAP039407; Procedure E-FP-08 Lacks Guidance for Loss of SRM Indication in Control Room; dated November 15, 2006

CAP039418; Incorrect location Identified for Appendix R Raceway JB1103; dated November 15, 2006

CAP039449; USAR Not Updated to Reflect Method of Evaluation in GL 96-06 Response; dated November 16, 2006

CAP039885; NRC Questions on Cable Spreading Room Lead Pipe; dated December 7, 2006

CAP040044; NRC Inspector Comment from Mod/50.59 Inspection; dated December 14, 2006

CAP040051; Technical Specification Question; dated December 14, 2006

CAP040053; NRC Question About Power Factors Listed in SP-42-047A and B; dated December 14, 2006

CAP040057; NRC Question Regarding 50.59 Compliance During Revision T of E-0-05; dated December 14, 2006

CAP040096; CSR Fire Suppression System Coverage - NRC Potential NCV of Appendix R, III.G.3; dated December 15, 2006

Corrective Action Program Documents Reviewed During the Inspection

CAP 027495; Service Water Supplies to CRAC Units Potentially Impacted by Tornado Missiles; dated May 20, 2005

CAP033687; Cable Spreading Room Fire Suppression System Adequacy Is Questioned; dated May 9, 2006

CAP 037591; RHR Flooding Mod Does Not Appear to Have Addressed All Components; dated September 25, 2006

CAP038824; CREZ Boundary Function Not Adequately Reflected in USAR Safety Analysis; dated October 26, 2006

CE017095; IPEOPs Use WR Cntmt Sump Indication, Contrary to USAR and NRC SER; dated February 28, 2006

Calculations

51-9014070; Kewaunee Strainer Performance Test Report; Revision 1

C10059; Aux Bldg. Cable Trays No.'s 1AT16S5; 1AT17S6; 1AT14S6; dated November 6, 1978

C11738; Evaluation of Flood Sources Inside the RHR; Revision 0

FPEE-049; Evaluation of Partial Area Suppression/Detection; Revision 3

X10072; Safe Shutdown Assessment of Internal Flood Levels Due to Postulated Pipe or Tank Rupture in the Controlled Area of the Auxiliary Building; Revision 0

Drawings

SK-S-3622-01; RHR Pit Flooding Project, DCR 3622, Plans; Revision 5

SK-S-3628-1-2; Tornado Missile Shield for CRAC Room SW Pipes; Revision B

Procedures

E-FP-08; Emergency Operating Procedure - Fire; Revision AT

SP-42-047A; Diesel Generator “A” Operational Test; Revision AD

SP-42-047B; Diesel Generator “B” Operational Test; Revision AF

Licensing Basis Documents

Updated Safety Analysis Report; Revision 19

Miscellaneous Documents

OPR-151; Emergency Diesel Generator 1A and 1B; Revision 2

OPR-106; Service Water System and Control Room Air Conditioning; Revision 1

LIST OF ACRONYMS USED

°	Degrees
ΔCDF	Difference in Core Damage Frequency
AF	Adjustment Factor
CAP	Corrective Action Process
CCDP	Conditional Core Damage Probability
CFR	Code of Federal Regulations
CRAC	Control Room Air Conditioning
DCR	Design Change Request
DF	Duration Factor
DPR	Demonstration Power Reactor
DRS	Division of Reactor Safety
F	Fahrenheit
F	Fire Frequency
FSAR	Final Safety Evaluation Report
GL	Generic Letter
GSI	Generic Safety Issue
IMC	Inspection Manual Chapter
IR	Inspection Report
LER	Licensee Event Report
NCV	Non-Cited Violation
NEI	Nuclear Energy Institute
NRC	U.S. Nuclear Regulatory Commission
OPR	Operability Recommendation
PORV	Power Operated Relief Valve
PNS	Probability of Non-Suppression
RHR	Residual Heat Removal
SDP	Significance Determination Process
SF	Severity Factor
SSCs	Structures, Systems, and Components
SW	Service Water
URI	Unresolved Item
USAR	Updated Safety Analysis Report